NON-PUBLIC?: N
ACCESSION #: 8812290147
LICENSEE EVENT REPORT (LER)

FACILITY NAME: CATAWBA NUCLEAR STATION UNIT 2 PAGE: 1 OF 6

DOCKET NUMBER: 05000414

TITLE: MANUAL REACTOR TRIP ON DECREASING STEAM GENERATOR LEVEL DUE TO DESIGN
DEFICIENCY
EVENT DATE: 11/23/88 LER #: 88-031-00 REPORT DATE: 12/22/88

OPERATING MODE: 1 POWER LEVEL: 048

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
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ASSOCIATE ENGINEER - LICENSING

COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE TO NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On November 23, 1988, at 0919:56 hours, an inadvertent momentary actuation of the Train A Interior Doghouse Level Switch initiated a signal for several valves to close. Two valves, 2CF51, S/G 2C CF CONT ISOL and 2CF10, CF Pump 2A DISCH ISOL, initiated closure, but 2CF51 reopened after the signal cleared. 2CF10 continued to close because of seal-in logic in the circuitry. As the valve closed, Main Feedwater Pump Turbine (CFPT) 2A tripped on high discharge pressure. CF Pump 2B was out of service for maintenance. Consequently, a Main Turbine Trip and an Auxiliary Feedwater Autostart on trip of both CFPTs were initiated. Control Room Operators initiated a reset of CFPT 2A in an attempt to reestablish Feedwater flow, but rapidly decreasing Steam Generator levels caused the Operators to manually initiate a Reactor Trip. A Feedwater Isolation was initiated on Reactor Trip coincident with Low Tave. The Unit was operating at 48% power at the time of this incident.

The actuation of the level switch was attributed to K-Mac cleaning personnel

sweeping near the level switch and contacting the level switch displacer. This incident has been attributed to a design deficiency because the circuit design for the level switches and the mounting design for the standpipe were not adequate to prevent an inadvertent actuation of the High Doghouse Level Isolation circuitry. Design Engineering will analyze the potential benefits of additional level switches, interlocks and time delays for the circuits, and will also devise a protection mechanism for the level switch displacers. Training will be evaluated for all appropriate personnel concerning sensitivity of plant instruments and controls. The health and safety of the public were unaffected by this event.

END OF ABSTRACT

BACKGROUND:

The High Doghouse Level interlock is designed to protect important safety functions of valves EIIS:V! located in the Doghouses, such as Feedwater Isolation, Main Steam Isolation, and Containment Isolation, from flooding. In the event of a High Energy Line Break in the Doghouse, rising water level would actuate the level switches at 11 inches increasing and close all valves which would be likely sources of water to the Doghouse. Additionally, the interlock would trip both Feedwater Pumps EIIS:P! and close both Feedwater Pump Discharge Valves.

DESCRIPTION OF INCIDENT:

On November 23, 1988, at approximately 0830 hours, while operating at 48% power, a K-Mac cleaning crew entered Unit 2 Interior Doghouse for routine cleaning. After several exits and re-entries for materials and lighting, a crew of four personnel remained to perform the cleanup. A K-Mac employee worked his way to the area near the Interior Doghouse Level switches at approximately 0910 hours. At this time he moved scaffolding materials from the area near the Train A Interior Doghouse Level switch, 2CFLS6000. In doing so, he bumped the junction box for termination of wires from the level switch to the main cable. This bumping apparently had no effect on the level switch.

Approximately ten minutes later, while using a short bristle brush to sweep the area, the employee cleaned under the level switch standpipe, unknowingly contacted the displacer (which was partially exposed beneath the standpipe) and moved the displacer upwards. At 0919:56 hours, the level switch actuated the INNER DOGHOUSE TRAIN A LEVEL HIGH annunciator and initiated closure for the associated valves. Because of the short duration of the high level signal, the alarm cleared rapidly. 2CF51, S/G C CF Containment Isolation, which had started to close, returned to its open position. 2CF10, Main Feedwater EIIS:SJ!

(CF)
Pump 2A Discharge Isolation, which had also started to close, could not return to its open position because of seal-in circuitry. As the valve motor EIIS:MO! actuator continued to close the valve, CF Pump 2A Discharge Pressure increased. Due to throttling action of the valve, Main Feedwater EIIS:SJ! flow decreased and created a Steam Flow/Feedflow mismatch. The CF Pump Turbine EIIS:TRB! speed increased automatically in an attempt to correct the mismatch. CF Pump 2A Discharge Pressure increased to the High Discharge Pressure Trip setpoint of 1385 psig, reached a maximum of 1398 psig, and tripped the pump at 0922:23 hours. CF Pump 2B was previously isolated for maintenance while operating at approximately 50% power, so the loss of both feedwater pumps initiated a Main Turbine Trip and Auxiliary Feedwater EIIS:BA! (CA) autostart. An automatic Reactor EIIS:RCT! runback was initiated, Blowdown Isolation automatically actuated, and the Main Steam EIIS:SB! Bypass (SB) valves modulated open to control Tave.

At 0922.36 hours, 2CF10 had reduced flow through the CF Pump to approximately 2000 gpm, actuating the low flow alarm. The valve finished closing at 0922:37 hours.

2CF6, CF Pump 2A Recirc Control Valve, opened at 0922:43 hours, on low pump suction flow (after CFPT 2A had tripped) and cleared the alarm.

The CROs reset CFPT 2A and attempted to restore feedwater flow, but because of rapidly decreasing Steam Generator EIIS:SG! (S/G) levels, the CROs manually initiated a reactor trip at 0923:09:407 hours.

Less than four seconds later, the first two Steam Generator Low Level alarms actuated for S/G B. At 0923:14:639 hours, the second low level alarm for S/G D was actuated. Low level in 2 of 4 S/Gs initiated a Turbine Driven Auxiliary Feedwater Pump (TDCAP) autostart, and both steam supply valves opened.

At 0923:42 hours, Reactor Coolant EIIS:AB! Tave decreased to 564 degree F and initiated a Feedwater Isolation on Reactor Trip with Low Tave. All appropriate isolation valves indicated closed with the exception of 2CF60, S/G 2D CF Containment Isolation.

At approximately 0925:40 hours, the CROs secured Hotwell Pump C and Condensate Booster Pump C because of diminished Feedwater flow requirements.

At approximately 0926 hours, S/Gs levels started to increase from 50% wide range level. At that time, CROs secured the TDCAP and throttled CA flows to the S/Gs.

Upon noticing that 2CF60 was not indicating closed, an Operator was dispatched to verify the valve position. He reported that the valve was closed but the closed limit switch was not actuated. An investigation of the alarm on High Doghouse level revealed the presence of K-Mac personnel cleaning near the level switch and determined the cause of the incident.

At approximately 1030 bours, 2CF10 was reopened by the CROs. At approximately 1120 hours, the Blowdown Isolation valves were returned to their original positions.

CROs attempted to cycle 2CF60 several times without success. The valve did not cycle fully. A high priority work request, 42017 OPS, was initiated to repair the problem. CROs realigned the valves affected by the Feedwater Isolation signal and aligned CF pump 2A to supply Feedwater to the S/Gs. They then secured CA Pumps 2A and 2B. Instrumentation and Electrical(IAE) personnel investigated the actuator for 2CF60 and determined that it required replacement.

Auxiliary Feedwater Pump 2B was restarted and CA flow reestablished for S/G D while 2CF60 was isolated for repairs. The actuator was replaced and satisfactorily retested on November 24, at 2330 hours.

While repairs were being performed on 2CF60, the operability of 2CF42, S/G B Containment Isolation, was evaluated. The valve received the same signal to close as 2CF51 but no position change was noted on the valve during the incident. Subsequently, a test was conducted by Performance personnel to determine the ability of 2CF42 to properly respond to the High-High Doghouse Level signal. 2CF42 was successfully cycled and performed within specifications.

During restart, problems with 2CF6, CF Pump 2A Recirc control, caused an automatic initiation of Auxiliary Feedwater (see LER 414/88-32).

The Unit returned to Mode 1, Power Operation, at 0952 hours, on November 25, 1988.

CONCLUSION:

This incident has been attributed to a design deficiency for several reasons. The Doghouse level switch standpipes were not designed for placement in a normally dry area which would require periodic cleaning. The standpipe does not protect the level switch displacers from inadvertent actuation. Because of the single actuation design of the High Doghouse Level Isolation circuitry, any instantaneous actuation of one level switch would cause trips on both CFPTs due to closure of both CF Pump discharge valves. Station personnel requested the deletion of this interlock in 1986. It was determined that a level switch actuation was unlikely and therefore did not warrant its removal. Additionally, Technical Specification requirements for the inoperability of any single level switch requires immediate action to shut the Unit down. This design does not enhance Unit reliability or availability.

A search of the Operating Experience Program database revealed approximately 30 previous incidents in which an ESF actuation was attributed to a Design Deficiency. However, there have been no previous actuations of the High Doghouse Level signal at Catawba. Therefore, this incident is not considered to be recurring.

CORRECTIVE ACTION:

SUBSEQUENT

(1) The Unit was stabilized in Mode 3.

(2) Station personnel determined the actuation to be due to cleaning activities in the area.

(3) An Operator was dispatched to verify the position of CF60.

(4) Replacement of the actuator for 2CF60 was completed.

(5) Performance tested CF42.

(6) Operations submitted a Station Problem Report for Design Engineering to reanalyze the proper automatic plant response to a High Energy Line Break in a Doghouse, including the proper Feedwater Pump Turbine and Discharge valve responses, and protection for Doghouse level switches.

PLANNED

(1) A proposed training program to alert personnel to the sensitive and delicate nature of some plant equipment will be distributed for station management review.

(2) The failure of the actuator for 2CF60 to cycle after the trip will be evaluated.

(3) Corrective actions for all inadequate response items identified in the Post-trip Review will be developed.

SAFETY ANALYSIS:

Due to the trip of the operating CF pump, an automatic Turbine trip and CA Motor Driven Pump autostart occurred. The Rod Control EIIS:AA! System was in automatic mode at initiation of the pre-trip transient, and the Reactor was operating at 48% power, below the P-9 setpoint (69% power). Therefore, upon Turbine trip the Unit ranback from 48% power to 27% power in approximately 46 seconds. S/G levels began decreasing due to the trip of the CF Pump, and the Reactor was manually tripped at 27% power in anticipation of a S/G Low-Low Level Reactor Trip signal. A Turbine Driven CA Pump Autostart signal occurred approximately 6 seconds after manual Reactor trip upon low-low level in two-out-of-four S/Gs. The redundant steam supply valves for the Turbine Driven CA Pump, SA2 and SA5, opened within 3 and 6 seconds, respectively, of the autostart signal. A Feedwater Isolation occurred as designed upon Reactor trip with low Tave (564 degree F). All of the control rods fell to the bottom of the core, reducing power to decay heat level.

During the Unit runback, Reactor Coolant System temperature increased from 572 degree F to 578 degree F. Upon Reactor trip, Reactor Coolant temperature stabilized at 554 degree F, 3 degree from the desired no-load target of 557 degree F. During the Unit runback, Reactor Coolant System pressure increased from 2235 Psig to a maximum value of 2283 psig. Reactor Coolant System pressure decreased to a minimum value of 2076 psig post-trip, and stabilized at the no-load target of 2235 psig within 15 minutes, which was well within system allowable post-trip parameters. Pressurizer level increased from 41% to a maximum value of 49% during the runback. Pressurizer EIIS:PRZ! level decreased to a minimum value of 25% post-trip, and then stabilized at 27% within 30 minutes post-trip, 2% from the no-load target of 25%. Steam pressure increased to a maximum value of 1128 psig upon Turbine trip and Unit runback. Steam pressure decreased to a minimum value of 1040 psig post-trip, and then stabilized at 1055 psig within 30 minutes post-trip, 35 psi from the no-load target of 1090 psig. S/G C Narrow Range Level decreased to a minimum value of 4% post-trip. S/Gs A, B, and D Narrow Range indication dropped off-scale post-trip. S/Gs A, B, C, and D levels reached a minimum wide range indicated value of 51%, 47%, 50%, and 48%, respectively. Steam pressure correction of these values yields actual levels of 68%, 61%, 66%, and 64% for S/Gs A, B, C, and D, respectively, which is within acceptable parameters.

With the exception of valve SB27, Main Steam Bypass to the Condenser EIIS:COND!, which was isolated for maintenance, all three banks of Steam Dump to Condenser Valves opened during the Unit runback. Banks 1 and 2 remained open to dump steam to the Condenser to appropriately control Tave during the post-trip response. The S/G ORVs were not required to open to dump steam, as the maximum steam pressure was within the allowable setpoint tolerance. The Operators manually actuated Pressurizer Heater EIIS:HTR! Bank B within 15 seconds post-trip to limit pressurizer pressure decrease. Auxiliary Feedwater flow was above the 450 gpm minimum value required by the Reactor Trip Emergency

Procedure, and within approximately four minutes was throttled by the Operators to prevent excessive cooldown. The Reactor Coolant was 86 degree F subcooled at the point of minimum Reactor Coolant System pressure. Adequate heat sink for core decay heat removal was available and maintained at all times.

Section 15.2.7 of the Catawba FSAR assumes credit for a) Reactor trip upon S/G Low-Low Level Signal, b) CA autostart one minute after S/G Low-Low Level Signal, and c) availability of the S/G PORVs and Code Safety Valves to dissipate residual heat. In this event the Reactor was manually tripped 5 seconds prior to when the

first S/G Low-Low Level Trip Signal was automatically generated. Also, since offsite power was available, the Auxiliary Feedwater Pumps were autostarted when the signal was received. The Steam Dump to Condenser Valves, S/G PORVs, and S/G Code Safety Valves were available to dissipate residual heat if necessary. Therefore, this event is fully bounded by the "Loss of Normal Feedwater Flow" transient as described in Section 15.2.7 of the Catawba FSAR.

The cooldown limits of 100 degree F pe hour for the Reactor Coolant System and 200 degree F per hour for the Pressurizer were not exceeded. Integrity of the Fuel Cladding, Reactor Coolant System, and Containment structure was maintained at all times.

After the CF Isolation signal occurred, CF60, S/G D Containment Isolation Valve, did not indicate closed. Local observation verified that the valve appeared to be closed. It is difficult to verify with conclusive certainty whether the valve stoked fully closed or partially closed. However, it is also likely, based on local observation, that even if the valve did not stroke to the fully closed position, it travelled the majority of its stroke distance. Heat and mass input to Containment and cooldown effects of flow through this valve in certain postulated scenarios would only be a concern if the CF Pump trip circuitry did not function properly (as it is not safety grade circuitry) and if the valve (CF60) was far enough open to pass substantial flow. Even if the CF Pump trip function did not function properly, the small amount of flow passed through this valve could be isolated by manual closure of valve 2CF55, S/G D Feedwater Control Valve, which has a handwheel operator.

Based on the preceeding analysis, it may be concluded that the health and safety of the public were not affected by this event.

This event is reportable pursuant to 10 CFR 50.73 (a)(2)(iv).

ATTACHMENT 1 TO 8812290147 PAGE 1 OF 1

DUKEPOWER

December 22, 1988

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Catawba Nuclear Station, Unit 2
Docket No. 50-414
LER 414/88-31

Gentlemen:

Pursuant to 10 CFR 50.73 Section (a) (1) and (d), attached is Licensee Event Report 414/88-31 concerning a manual reactor trip on decreasing steam generator level due to a design deficiency.

This event was considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

Hal B. Tucker

JGT/IIR88

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